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10 CFR 50.90

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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

H.B. Robinson Steam Electric Plant, Unit 2
Renewed Facility Operating License No. DPR-23
Docket No. 50-261

Subject: Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"

References:

1. Duke Energy letter, *Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"*, dated April 5, 2018 (ADAMS Accession No. ML18099A130).

2. Duke Energy letter, *Supplement to the Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"*, dated June 6, 2018 (ADAMS Accession No. ML18162A147).

3. NRC E-Mail, *Robinson RAIs – LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L 2018-LLA-0095) and LAR to Adopt TSTF-425 (EPID L 2018-LLA-0104)*, dated October 12, 2018 (ADAMS Accession No. ML18288A019).

Ladies and Gentlemen:

By letter dated April 5, 2018 (Reference 1), as supplemented by letter dated June 6, 2018 (Reference 2), Duke Energy Progress, LLC (Duke Energy) submitted a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2). The proposed amendment would modify the licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors."

By correspondence dated October 12, 2018 (Reference 3), the Nuclear Regulatory Commission (NRC) staff requested additional information from Duke Energy that is needed to complete the LAR review.

The enclosure to this letter provides Duke Energy's response to the NRC RAI. Attachment 1 contains PRA implementation items which must be completed prior to implementation of 10 CFR 50.69 at HBRSEP2. Attachment 2 contains proposed markups of the HBRSEP Renewed Facility Operating License.

The conclusions of the original No Significant Hazards Consideration and Environmental Consideration in the original LAR are unaffected by this RAI response.

There are no regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

Should you have any questions concerning this letter and its enclosure, or require additional information, please contact Art Zaremba at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 13, 2018.

Sincerely,

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JLV

Enclosure: Response to NRC Request for Additional Information cc:

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Serial: RA-18-0193

H.B. Robinson Steam Electric Plant, Unit No. 2
Docket No. 50-261 / Renewed License No. DPR-23

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors”

Enclosure

Response to NRC Request for Additional Information

NRC Request for Additional Information

By letter dated April 5, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18099A130), as supplemented by letter dated June 16, 2018 (ADAMS Accession No. ML18162A147), Duke Energy Progress, LLC, (Duke Energy, the licensee), submitted a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2). The proposed amendment would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," and provide the ability to use probabilistic risk assessment (PRA) models, namely the internal events PRA (IEPRA), internal flooding PRA (IFPRA), and internal fire PRA (FPRA) for the proposed 10 CFR 50.69 categorization process.

Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (ADAMS Accession No. ML061090627), endorses, with regulatory positions and clarifications, the Nuclear Energy Institute (NEI) guidance document NEI 00-04, Revision 0, "10 CFR 50.69 SSC [Structure, System, and Component] Categorization Guideline," July 2005 (ADAMS Accession No. ML052910035), as one acceptable method for use in complying with the requirements in 10 CFR 50.69. Both RG 1.201 and NEI 00-04 cite RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004 (ADAMS Accession No. ML040630078), which endorses industry consensus PRA standards, as the basis against which peer reviews evaluate the technical acceptability of a PRA. Revision 2 of RG 1.200 issued March 2009 is available at ADAMS Accession No. ML090410014.

Section 3.1.1 of the LAR states that Duke Energy will implement the risk categorization process of 10 CFR 50.69 in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. However, the licensee's LAR does not contain enough information for the U.S. Nuclear Regulatory Commission (NRC) staff to determine if the licensee has implemented the guidance appropriately in NEI 00-04, as endorsed by RG 1.201, as a means to demonstrate compliance with all of the requirements in 10 CFR 50.69, including technical adequacy of the PRA models. The NRC staff requests additional information (RAI) for the following areas in order to complete its assessment.

The NRC staff notes that by letter dated April 16, 2018 (ADAMS Accession No. ML18117A006), Duke Energy submitted a LAR for HBRSEP2 to adopt Technical Specifications Task Force (TSTF) 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b." RAIs regarding the HBRSEP2 LAR to adopt TSTF-425 are in Enclosure 2. The RAIs for this LAR, to implement 10 CFR 50.69, are similar in nature to the RAIs for HBRSEP2 to adopt TSTF-425. As such, the NRC staff requests separate responses to the RAIs for the HBRSEP2 LAR to implement TSTF-425 LAR and the HBRSEP2 LAR to implement 50.69, even though the responses may be similar.

PRA RAI 01 - Open/Partially Open Findings in the Process of Being Resolved:

Section 4.2 of RG 1.200 states that the LAR should include a discussion of the resolution of the peer review facts and observations (F&Os) that are applicable to the parts of the PRA required for the application. This discussion should take the following forms:

- A discussion of how the PRA model has been changed and
- A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.

Attachment 3 of the LAR, "Disposition and Resolution of Open Peer Review Findings and Self- Assessment Open Items," provides finding-level F&Os that are still open or only partially resolved after the F&O closure review. Also, F&O descriptions and their dispositions were previously provided to the NRC in the LAR to adopt for Technical Specification 5.5.16 Option B, "10 CFR 50, Appendix J, Integrated Leak Rate Test Interval and Type C Leak rate testing Frequency" (ADAMS Accession No. ML16201A195) and in the LAR to adopt National Fire Protection Association Standard 805 (ADAMS Accession No. ML16337A264). For a number of F&O dispositions there is insufficient information for NRC staff to conclude that the F&O is sufficiently resolved for this application.

a. <u>F&Os associated with Supporting Requirements (SR) AS-A5, AS-B3, LE-C4, and LE-D5</u> regarding thermally induced steam generator tube ruptures (SGTR):

   Open F&Os on SRs AS-A5, AS-B3, and LE-D5 in the LAR state, in part, that the thermally induced SGTR accident sequence was missing from the PRA. Separately, the LAR supplement indicates that the F&O associated with SR LE-D6 (SR LE-D6 directs that a thermally induced SGTR shall be modelled) was closed because a thermally induced SGTR accident sequence was developed and peer-reviewed with no subsequent F&Os. However, the resolutions for the open F&Os, associated with SRs LE-C4 and LE-D5 in the LAR also states, in part, that a sensitivity study demonstrates that un-modelled human failure events (HFEs) related to isolating a ruptured SG following an SGTR initiating event (i.e., apparently not a thermally induced SGTR) has a minimal impact on the PRA results and an acceptable impact of the 50.69 categorization.

   i. Clarify if the evaluation of the impact of the un-modelled isolation HFE described in the F&O resolution for SRs LE-C4 and LE-D5 in the LAR include the thermally induced SGTR accident sequence. If not please include the thermally induced SGTR accident sequence in the sensitivity study or otherwise evaluate its impact.
   ii. Provide clarification that the sensitivity study related to the exclusion of the SG isolation HFE demonstrated that there was no impact on any SSC risk categorization, or

   iii. Alternatively to Part ii, if the sensitivity study demonstrates that the exclusion of the operator action does impact any risk categorization, then propose a mechanism to ensure incorporation of the operator action in the PRA model of record (MOR) prior to implementation of the 10 CFR 50.69 categorization program.

Duke Energy Response to PRA RAI 01.a.:

The evaluation of the impact of the un-modelled isolation HFE described in the F&O resolution for SRs LE-C4 and LE-D5 in the LAR does not apply to the thermally induced SGTR accident sequences. The referenced HFE addresses isolating feed flow to, and steam flow from, a SG that has had a tube rupture as the initiating event, in order to allow equalizing primary and secondary side pressure to stop flow out the break. It is only applicable to SGTR initiating events. Thermally induced SGTRs occur following core damage, given that the secondary side of the SG is faulted, such that there is little or no opportunity for operators to isolate the ruptured generator to prevent LERF. Therefore, credit for preventing LERF due to induced SGTRs by isolating a faulted SG is not taken.

A sensitivity study has been performed to assess the impact of adding an HFE to the model for isolating a ruptured SG during an SGTR initiating event. The HFE was developed and dependency between it and the other HFEs in the model was assessed. The results of the sensitivity study showed that several PRA basic events exceeded the Fussell-Vesely or Risk Achievement Worth importance value thresholds for high safety-significance in 10 CFR 50.69 with the new HFE in the model which did not exceed the threshold with the HFE excluded. Therefore, this operator action will be included in the RNP PRA model prior to implementing 10 CFR 50.69. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing 50.69.

b. <u>F&O associated with SR IFEV-A7-01</u> regarding human-induced flood events:

One of the issues provided in the F&O description in LAR Attachment 3 concerns the proper screening of human-induced flood events to determine exclusion from the PRA MOR. The first part of the disposition states, “[t]he sensitivity study performed was overly conservative and attempted to apply all industry human induced failure events on a per piping frequency. This led to a largely over conservative value.” There is no description or results for this sensitivity study provided in the LAR.

The second part of the disposition states, “[h]man induced flooding events are not risk significant for this application as on the whole human induced flooding events in the industry have largely been occurring less often.” The disposition makes reference to the period from 1971 to 2011, which appears to match the period used in the EPRI TR-3002000079, “Pipe Rupture Frequencies for Internal Flooding Probabilistic Assessments,” Revision 3, which provides flood event probabilities including human-induced events. The NRC staff notes that the EPRI TR is an update of the 2006 TR data and would reflect the decreasing trend of events over that period.

The NRC staff has issued two information notices (IN) related to human-induced flooding events since 2007, IN 2007-01 and IN 2016-11.

Section 5.6 of EPRI TR-1019194, “Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment,” provides specific methodology, including screening, for maintenance-induced flooding events.

Capability Category (CC) I/II for ASME/ANS 2009 PRA Standard for SR IFEV-A7 states, “[i]nclude consideration of human-induced floods during maintenance through application of generic data.” SRs IFSN-A10 and IFSN-A15 provide flood event screening criteria. In light of these observations:

i. Describe the sensitivity study mentioned in the F&O disposition. Include in this discussion the purpose of the sensitivity study, what modifications to the PRA model were performed, the results of the study, and the insights from this sensitivity study.

ii. Provide justification, such as industry approved screening criteria, to exclude the remaining maintenance-induced internal flooding events, using industry generic data, from the PRA model, and provide justification that exclusion of these maintenance- induced events does not affect SSC risk categorization, or

iii. Alternatively to Part ii, propose a mechanism to ensure F&O IFEV-A7-01 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism
should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.b.:

The sensitivity study referred to in the LAR apportioned the human induced flood frequency for all generic human induced flood scenarios by allocating the human induced flooding frequency by the amount of piping for that system in the flood area divided by the total amount of system piping in the plant. It then added this failure frequency to the model for the scenarios and quantified the result. It made no attempt in screening or qualifying whether maintenance induced flooding was possible in the specific system or flood area. This had the impact of dramatically overstating the impacts of human induced flooding and increasing CDF and LERF values (44% and 55% delta to CDF and LERF respectively), but not providing any real risk insight in terms of the impact human induced flooding.

The generic human induced flooding data will be reviewed to determine whether any industry maintenance induced flooding events need to be added to the RNP IFPRA. This will be accomplished by reviewing the generic data from EPRI TR 3002000079 and screening out events that are not applicable to RNP. Following this, maintenance practices and activities at RNP will be reviewed to determine whether the generic maintenance event is applicable. Any of the generic human induced flood events that are not screened out based on this process will be included in the RNP IFPRA. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing 10 CFR 50.69.

c. F&O associated with SR IFSN-A8-01 regarding door failure heights

The description of the finding in LAR Attachment 3 states the, "[u]se of EPRI door failure criteria of 1 ft [foot] / 3 ft may not be appropriate depending on the actual door attributes and flooding scenario."

The disposition states, "[t]he current IFPRA assumes that the majority of the components would fail at or around 1 ft to 3 ft," and concludes the effects, "minimal on modeling results and therefore will have no impact on the quantified values with regard to the 50.69 application." The disposition does not discuss how the application provides a bounding assessment for this assumption.

Appendix D of EPRI TR-1019194, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment," provides methodology for determining door failure heights. In light of these observations:

i. Provide justification, such as a sensitivity study, that the exclusion of the correct door failure heights would not impact any SSC risk categorization, or

ii. Alternatively, propose a mechanism that ensures F&O IFSN-A8-01 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.c.:

The 1 ft or 3 ft door failure height is taken from the EPRI Internal Flooding Guidelines (EPRI TR-1019194). This value is used in lieu of a plant specific value. This is a simplifying assumption for the IFPRA. The only aspect of the IFPRA impacted by this assumption is the time available for operators to isolate the flood source prior to the door failing, which impacts the probability of isolation failure. The difference between the generic failure height and a door-specific failure height is expected to be very small since the doors at RNP are fairly typical of nuclear power plant doors. Small differences in failure height would lead to small differences in the time available for isolation which would then lead to a negligible difference in the calculated isolation failure probability. Therefore, the value used from the EPRI Internal Flooding Guidelines is reasonable.

As discussed in the response to PRA RAI 03, uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria) are being addressed by the sensitivity study required by section 8.1 of NEI 00-04 and performance monitoring of low safety significant (LSS) components as required by 10 CFR 50.69(e)(3). As discussed in PRA RAI 03, the sensitivity and monitoring program are sufficient to address model uncertainties and assumptions not related to components which were excluded from the model. Since the door height failure assumption only has the potential to affect operator action timing, which can affect flood operator failure probabilities and impact scenario risk values, it did not result in any components being excluded from the model. Therefore, this assumption is appropriately addressed by the sensitivity study and performance monitoring program.

d. F&O associated with SR IFSN-A8-02 regarding door gap flooding propagation

The disposition in LAR Attachment 3 states that it identified one scenario where additional equipment would be impacted. In evaluating the additional failures the disposition states, "[c]rediting flow underneath door gaps would increase the time that operators would be able to potentially isolate the scenario. Therefore as it is currently modeled, scenarios for this flood area are conservative." The disposition concludes, "[t]he timing effects of this open F&O is minimal on modeling results and therefore will have no impact."

In accordance with the SR IFSN A10 (ASME/ANS 2009 PRA Standard), each developed flood scenario includes, "giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs." The NRC staff notes the exclusion of SSC(s) impacts from initiating events reduces their contribution to risk and can therefore impact their importance measures, thus potentially impacting the importance measures of other SSC(s) as well. In light of these observations:

i. Provide justification, such as a sensitivity study, that the exclusion of the additional PRA SSC impacts from the door gap propagation has no impact on the 10 CFR 50.69 categorization results, or

Alternatively, propose a mechanism that ensures F&O IFSN-A8-02 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.d.:

The only important flood area in the RNP IFPRA that has substantial door gaps found during the walkdown is a large open area that would not allow for water to accumulate to a large depth. In addition, the flood area is abutted by double doors that open out from the flood area. This in effect would preclude the accumulation of water and for a large driving hydrostatic head to develop. As shown in the IFPRA analysis RNP door gaps are in general small and a large depth of water would need to develop to induce a significant amount of water via inter door propagation. Treatment of door gaps is a model uncertainty in any IFPRA. Door gap treatment in the RNP IFPRA is found to be reasonable and in line with industry practice on the evaluation and subsequent treatment.

As discussed in the response to PRA RAI 03, uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria) are being addressed by the sensitivity study required by section 8.1 of NEI 00-04 and performance monitoring of low safety significant (LSS) components as required by 10 CFR 50.69(e)(3). As discussed in PRA RAI 03, the sensitivity and monitoring program are sufficient to address model uncertainties and assumptions not related to components which were excluded from the model. Since the flood propagation through door gaps assumption does not result in any components being excluded from the model, this assumption is appropriately addressed by the sensitivity study and performance monitoring program.

PRA RAI 02 – Qualitative Function Categorization:

Table 3-1 of the LAR indicates that the evaluation of the seven qualitative criteria defined in Section 9.2 of NEI 00-04 is performed at the function level and prior to the Integrated Decision-making Panel (IDP). The LAR states that “NEI 00-04 only requires the seven qualitative criteria in Section 9.2 of NEI 00-04…. to be completed for components/functions categorized as LSS.” LAR Table 3-1 Table 1 contains the entry “Allowable” at the intersection of the “IDP change HSS [high safety significant] to LSS [low safety significant]” column and “Qualitative Criteria” row, which appears to contradict the premise that the seven criteria are only applied to LSS functions. The guidance in NEI 00-04 states that the IDP “should consider the impact of loss of the function/structure, system, and component (SSC) against the remaining capability to perform the basic safety functions.”

Explain how the IDP will collectively assess the seven specific questions to identify a function/SSC as LSS as opposed to HSS including a clarification of the “Allowed” entry in LAR Table 3-1 and confirm that a negative answer to any of the seven questions would result in the function/SSC to be categorized as HSS.

Duke Energy Response to PRA RAI 02:

The assessments of the qualitative considerations are agreed upon by the Integrated Decision-making Panel (IDP) in accordance with NEI 00-04 Section 9.2. It is generally expected that a 50.69 categorization team will provide preliminary assessments of the seven considerations for the IDP’s consideration, however this is not a requirement and the final assessments of the seven considerations are the direct responsibility of the IDP.

In cases where the 50.69 categorization team provides a preliminary assessment of the seven qualitative considerations to the IDP, the seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e. all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

PRA RAI 03 – Identifying Key Assumptions and Uncertainties that Could Impact the Application:

Section 1.3 of RG 1.200 describes the level of detail of a PRA required and states, "[i]n general, the level of detail for the base PRA needs to be consistent with current good practice." Current good practices are those practices that are generally accepted throughout the industry and have shown to be technically acceptable in documented analyses or engineering assessments.

Section 3.2.7 of the LAR states that, "[t]he detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Revision 0) ["Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (ADAMS Accession No. ML090970525)] and Section 3.1.1 of EPRI TR- 1016737." The NRC staff notes that one of these sources has been superseded by a revision (Revision 1 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (ADAMS Accession No. ML17062A466), which references the updated EPRI guidance TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," (2012)).

Attachment 6 of the LAR contains ten key assumptions/sources of uncertainties from three PRA models, whereas industry guidance documents such as NUREG-1855, Revision 1, and EPRI TR-1026511 address a large number of potential assumptions and uncertainties. For example two key sources of fire PRA modeling assumptions/uncertainty are provided in the LAR, compared to the 2012 EPRI document which identifies 71 potential sources of uncertainty. There appear to be no uncertainties or assumptions associated with large early release (LERF) and internal flooding.

The LAR continues, "[t]he list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application."

The NRC staff notes that Stages C, D, E, and F of NUREG-1855 (Revision 1) provides guidance on how to identify key sources of uncertainty relevant to the application.

To address the observations above, the staff requests the following additional information:

a. Provide a detailed summary of the process used to determine the impact of each of the 71 potential sources of uncertainty in the EPRI documents and describe how this process resulted in the final set of ten key assumptions and sources of uncertainty presented in Attachment 6 of the LAR. Include in this discussion an explanation of how the process is in accordance with NUREG-1855, Rev. 1, or another NRC-accepted method.

Duke Energy Response to PRA RAI 03.a.:

Step E-1 (section 7.2) of NUREG 1855, Revision 1 provides guidance for identifying and characterizing those sources of model uncertainty and related assumptions in the PRA required for the application.

Substep E-1.1 of the NUREG recommends using the detailed guidance and a generic list of sources of model uncertainty and related assumptions in EPRI 1016737 for the internal event hazard group (including LERF), and using the examples of sources of model uncertainty for the internal fires, seismic, Low Power Shutdown and Level 2 hazard groups in EPRI 1026511. For RNP, this process was performed by reviewing PRA documentation for generic issues identified in Table A.1 of EPRI 1016737, as well as identifying plant-specific assumptions and uncertainties, and is therefore consistent with step E-1.1 of the NUREG. EPRI 1026511 was not explicitly used to identify generic uncertainties in models other than the internal events model. However, of the models addressed by EPRI 1026511, only the RNP fire PRA is being used to support the current application.

Substep E-1.2 of NUREG 1855, Revision 1 involves identifying those sources of model uncertainty and related assumptions in the base PRA that are relevant to an application. Those that are irrelevant can be screened from further discussion. However, since this application uses the internal events, internal flood, and fire PRA models for both CDF and LERF, all model uncertainties and related assumptions identified for these models are considered relevant. The original process screened some based on other factors, which is not consistent with the latest version of the NUREG.

Substep E-1.3 of NUREG 1855, Revision 1 involves characterizing the identified sources of model uncertainty and related assumptions. This characterization involves understanding how the identified sources of model uncertainty and related assumptions can affect the PRA. For the RNP uncertainty analysis, this was performed for all identified uncertainties/assumptions.

Substep E-1.4 is a qualitative screening process that involves identifying and validating whether consensus models have been used in the PRA to evaluate identified model uncertainties. As stated in NUREG 1855, Rev. 1, the use of a consensus model eliminates the need to explore an alternative hypothesis. For the RNP uncertainty analysis, some uncertainties/assumptions were screened based on their use of a consensus method, however, others were screened based on additional criteria, which again is not entirely consistent with the NUREG.

Once all relevant uncertainties/assumptions are identified in Step E-1, Step E-2 (section 7.3) of NUREG 1855, Rev. 1 provides guidance for identifying those sources of model uncertainty and related assumptions that are key to the application. The input to this step is the list of the relevant sources of model uncertainty identified in Step E-1. These sources of model uncertainty and related assumptions are then quantitatively assessed to identify those with the potential to impact the results of the PRA such that the application's acceptance guidelines are challenged. This assessment is made by performing sensitivity analyses to determine the importance of the source of model uncertainty or related assumption to the acceptance criteria or guidelines. In the RNP uncertainty analysis, this step was performed qualitatively to arrive at the list of uncertainties, not quantitatively, and therefore is not entirely consistent with the NUREG. For those uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria), Stage F (section 8) of NUREG 1855, Rev. 1, provides guidance on justifying the strategy used to address the key uncertainties that contribute to risk metric calculations that challenge application-specific acceptance guidelines. This portion of the NUREG was not addressed in the original RNP uncertainty analysis.

b. If the process of identifying key sources of uncertainty or assumptions for these PRA models cannot be justified, provide the results of an updated assessment of key sources of uncertainty or assumptions.

Duke Energy Response to PRA RAI 03.b.:

The process for identifying sources of uncertainty and assumptions is described and compared to the process outlined in NUREG-1855 rev. 1, in the response to item a. This comparison shows that the initial RNP identification of sources of model uncertainties and related assumptions was consistent with Substep E-1.1 of the NUREG, with the exception that generic sources of uncertainties for the fire PRA identified in EPRI 1026511 were not explicitly reviewed. However, the process to assess the identified uncertainties/assumptions was not entirely consistent with all portions of the latest revision of the NUREG. As such, an updated assessment was performed, as described below.

Since the ultimate goal in assessing model uncertainty is to determine whether (and the degree to which) the risk metric results challenge or exceed the quantitative acceptance guidelines for the application, due to sources of model uncertainty and related assumptions, the first step in the updated evaluation was to identify the risk metrics used as acceptance guidelines for the 10 CFR 50.69 categorization process. For 10 CFR 50.69 categorization, the acceptance guidelines are actually threshold values for Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) for each SSC being categorized, above which the SSC is categorized as high safety significant (HSS), and below which the SSC is categorized as low safety significant (LSS). As described in Step E-2 of the NUREG, each relevant uncertainty/assumption requires some sort of sensitivity analysis, and each sensitivity performed to evaluate an uncertainty/assumption involves some change to the PRA results. Since any change to the PRA results has the potential to change the F-V and RAW importance measures for all components (SSCs), every relevant uncertainty/assumption has the potential to challenge the acceptance guidelines. That is, since RAW and F-V are relative importance measures, any change to any part of the model will generate a new set of cutsets and potentially impact the RAW and F-V for every SSC. Thus, the only way to evaluate the impact of a sensitivity is to quantify the sensitivity case and compare the F-V and RAW values for all SSCs against the base case F-V and RAW values to determine if any exceed the HSS threshold in the sensitivity case that did not previously do so.

However, as stated in Stage F of NUREG-1855 rev. 1 (section 8.1), an appropriate method for dealing with uncertainties and related assumptions that challenge or exceed the acceptance guidelines is to use compensatory measures or performance monitoring requirements. Section 8.5 of the NUREG states that performance monitoring can be used to demonstrate that, “following a change to the design of the plant or operational practices, there has been no degradation in specified aspects of plant performance that are expected to be affected by the change. This monitoring is an effective strategy when no predictive model has been developed for plant performance in response to a change”. Since no predictive model of the increase in unreliability following alternative treatment of LSS SSCs exists, this option is appropriate for 10CFR 50.69. In fact, the example of a performance monitoring approach to address key uncertainties/assumptions given in section 8.5 is the factor of increase sensitivity combined with the performance monitoring process described for 10CFR 50.69 in NEI 00-04. The NUREG states:

> One example of such an instance is the impact of the relaxation of special treatment requirements (in accordance with 10 CFR 50.69) on equipment unreliability. No consensus approach to model this cause-effect relationship has been developed. Therefore, the approach adopted in NEI 00-04 as endorsed in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," [NRC, 2006a] is to:
>
> • Assume a multiplicative factor on the SSC unreliability that represents the effect of the relaxation of special treatment requirements.
>
> • Demonstrate that this degradation in unreliability would have a small impact on risk.
>
> Following acceptance of an application which calls for implementation of a performance monitoring program, such a program would have to be established to demonstrate that the assumed factor of degradation is not exceeded.

The use of the sensitivity study required by section 8.1 of NEI 00-04 and performance monitoring of LSS SSCs as required by 10 CFR 50.69(e)(3) is appropriate to address key uncertainties and assumptions. The impact of any key uncertainty or assumption sensitivity would be to potentially cause an SSC to be categorized as HSS when the base PRA analysis showed it to be LSS. The potential impact of categorizing an SSC as LSS rather than HSS is that the SSC could have alternative treatments applied to it and as such, the possibility exists that the reliability of SSC could be reduced (i.e., the specified aspect of plant performance that is expected to be affected by the change is the reliability of the SSC). Per section 8.1 of NEI 00-04, a sensitivity is performed which assumes the unreliability of all LSS components is increased by a factor of 3 to 5. Since, as discussed in NEI 00-04, no significant decrease in reliability is expected, this is very conservative. Additionally, since the failure probability of all LSS SSCs are increased at the same time in the sensitivity, this approach addresses all uncertainties/assumptions which could potentially impact the LSS/HSS categorization. The LSS sensitivity then must be shown to demonstrate that even assuming this factor increase, the quantitative guidelines of Reg. Guide 1.174 are not exceeded. Thus, the LSS sensitivity demonstrates that the potential impact of all uncertainties/assumptions is acceptable. Additionally, a performance monitoring program must be established as part of the 10 CFR 50.69 process (per NEI 00-04 section 12) which will monitor the reliability of all LSS SSCs to ensure that the factor of increase assumed in the sensitivity is not exceeded. This ensures the validity of the sensitivity study following implementation.

It is noted that uncertainties/assumptions which are related to SSCs being excluded from the PRA model, either because they are not believed to be required for accident mitigation or because they perform a backup function to other equipment but were conservatively not credited in the model, may not be adequately addressed by the above sensitivity and performance monitoring program. If an SSC is not in the PRA model, but actually performs (or could perform) an accident mitigation function, and that SSC is categorized as LSS (based on non-PRA criteria) the factor increase sensitivity would not appropriately address the uncertainty associated with this assumption/uncertainty. This is because if there are no failure events in the PRA model for the SSC, the LSS sensitivity study has no events to which to apply the factor of increase. If, contrary to the assumption, the SSC is actually required for accident mitigation and has been included in the model, increasing its failure rate by the factor of increase could have an impact on the sensitivity results with respect to the RG 1.174 limits.

Based on the above discussion, an updated assessment of sources of uncertainty and assumptions was performed. All uncertainties and assumptions identified in the original RNP process consistent with Substep E-1.1 of NUREG-1855 rev. 1 (i.e., all identified internal events, internal flood, and fire, uncertainties/assumptions), and the generic sources of uncertainties for the fire PRA identified in EPRI 1026511 were reviewed to identify any that are not adequately addressed by the factor increase sensitivity study required by Section 8.1 of NEI 00-04 and the performance monitoring program required by Section 12 of NEI 00-04. The table below provides details of these uncertainties and their disposition for the 10 CFR 50.69 categorization process. All other relevant uncertainties and assumptions are adequately addressed by the factor increase sensitivity study and performance monitoring program. Due to the large number of uncertainties/assumptions addressed, these are not listed. The updated assessment of key sources of uncertainty and assumptions performed in response to this RAI supersedes the contents of Attachment 6 to the original LAR.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
1	Spurious transfer of motor-operated valves SI-867A/B is an insignificant contributor to SI system unavailability. These valves receive an "S" signal to open and are checked monthly in the control room to be in the correct position.	RNP/F-PSA-074, Table 1, Item 4.	The probability of an MOV transferring closed is 3.47E-08/hr. The likelihood of one of the two MOVs transferring closed over one month is 2.5E-05. Assuming a common cause factor of 0.1 (which is very conservative since common cause failures are not typically modeled for passive failure modes), the probability of both 867A and 867B transferring closed is 2.5E-06. The probability that both valves fail to re-open on receipt of an "S" signal to open is 6.7E-05, the overall probability of 867A and 867B transferring closed and failing to re-open is approximately 1.7E-10. This failure is equivalent to a failure of valves 870A and 870B to open. The common cause failure probability for 870A/B failing to open is 6.7E-05. Thus, this failure mode of injection flow is more than 2 orders of magnitude lower than other failures of injection flow, and it was excluded. This is a consensus method per supporting requirement SY-A15 of ASME/ANS RA-Sa-2009, such that this uncertainty does not need to be addressed further.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
2	Mispositioning SW supply and return valves to each SI pump (for cooling) is not a dominant contributor to SI unavailability. SW valves SW-512 through SW-517 are confirmed to be functional prior to the quarterly HHSI pump flow test.	RNP/F-PSA-074, Table 1, Item 6.	The probability of a manual valve transferring closed is 6.36E-09/hr. The likelihood of one of these valves transferring closed over one month is 4.7E-06. This failure is considered equivalent to a failure of the associated SI pump. The failure probability for an SI pump failing to start and run is 1.6E-03. Thus, this failure mode of an SI pump is more than 2 orders of magnitude lower than other failures of the pump, and it was excluded. This is a consensus method per supporting requirement SY-A15 of ASME/ANS RA-Sa-2009, such that this uncertainty does not need to be addressed further.
3	Hot leg recirculation is not modeled for High Head Safety Injection.	RNP/F-PSA-074, Table 1, Item 10.	Although the PRA model will not generate importance measures for the additional valves required for hot leg recirculation, if the HHSI system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these valves, or the valves will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
4	Flow diversion via the test lines is discounted. SI pumps A, B, and C are isolated from the test line by valves SI-932, SI-935, and SI-938 respectively. Valve SI-941 isolates the common return line. These valves are locked and double verified to be closed following monthly testing.	RNP/F-PSA-074, Table 1, Item 12.	The probability of a manual valve transferring open is 6.36E-09/hr. The likelihood of valve SI-932, SI-935, and SI-938 transferring closed over one month is 1.4E-05. The probability that valve SI-941 also transfers open is 4.7E-06. No common cause is considered since this is a passive failure mode and the valves are of different design. Thus, the failure of any of the test valves and the common return valve is 6.6E-11. This failure is considered equivalent to a failure of the associated SI pump. The failure probability for an SI pump failing to start and run is 1.6E-03. Thus, this failure mode of an SI pump is more than 2 orders of magnitude lower than other failures of the pump, and it was excluded. This is a consensus method per supporting requirement SY-A15 of ASME/ANS RA-Sa-2009, such that this uncertainty does not need to be addressed further.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
5	Operator failure to isolate the depressurized accumulators (potential nitrogen injection) is not included in LOCA modeling; this is judged unimportant with respect to overall LOCA model.	RNP/F-PSA-074, Table 1, Item 35.	Although the PRA model will not generate importance measures for the additional valves required to isolate the accumulators, if the passive injection system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these valves, or the valves will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
6	Failure of the respective flow element (FE), transmitter (FT) or controller (FIC) is assumed to be included in the failure rate for each AFW train's flow control valve (FCV).	RNP/F-PSA-074, Table 1, Item 43.	Although the PRA model will not generate importance measures for the flow components, if the AFW system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, since they support the function of the FCV, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
7	Failure of the auxiliary oil pump is included in the steam-driven pump failure data.	RNP/F-PSA-074, Table 1, Item 47.	Although the PRA model will not generate importance measures for the auxiliary oil pump, if the AFW system is categorized, appropriate surrogate PRA events will be used to generate importance measures for the pump, since it supports the function of the steam-driven pump, or the oil pump will be added to the model.
8	Valves in the steam-driven pump oil cooler backup cooling path are not credited in the model. These valves are not tested, therefore are being excluded from the model as a potential backup source of cooling to the SDP.	RNP/F-PSA-074, Table 1, Item 49.	Although the PRA model will not generate importance measures for the backup cooling components, if the system containing these components is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
9	No credit is given for AFW pump start by an AMSAC signal or by low voltage on 4kV Bus 1 or Bus 4. To simplify start logic modeling, only the most prevalent signals are modeled. Due to start signal reliability this underestimation is no concern.	RNP/F-PSA-074, Table 1, Item 55.	Although the PRA model will not generate importance measures for the components which generate these signals, if the system containing these components is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
10	Failure of the discharge check valve of one MDAFW pump to close in a two pump cross-tied system, failure of the other pump due to recirculation flow is not modeled. A check valve and the electro-hydraulic FCV would have to fail for one pump, coincident with a failure of the other pump to start or run. The discharge check valves are backseat tested once per quarter. Thus, this failure is considered to be probabilistically insignificant.	RNP/F-PSA-074, Table 1, Item 59.	The probability of a check valve failing to close is 1.04E-04/demand. The failure probability of a MDAFW pump to start or run is 2.7E-03. The probability of failure of the opposite pump due to recirculation flow is then 2.8E-07. This failure is equivalent to a failure of the non-failed MDAFW pump. Since the probability of failure of the MDAFW pump to start or run is 2.7E-03 from above, this failure mode of the MDAFW pump is more than 2 orders of magnitude lower than the failure probability of the pump itself, and it was excluded. This is a consensus method per supporting requirement SY-A15 of ASME/ANS RA-Sa-2009, such that this uncertainty does not need to be addressed further.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
11	Failure of the containment air recirculation (CARC) system cooling coils is considered negligible. Cooling coils are passive. Operating experience indicates coil failure is unlikely.	RNP/F-PSA-074, Table 1, Item 60.	Although the PRA model will not generate importance measures for the cooling coils, if the CARC system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these coils, since they support the function, or the coils will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
12	No credit is taken for the normally isolated SW Booster Pump suction cross-connect. Further development could be made in the future if deemed important.	RNP/F-PSA-074, Table 1, Item 63.	Although the PRA model will not generate importance measures for the suction cross-connect, if the system containing these components is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, since they support the function, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
13	Loss of instrument air (IA) supply to the SG PORV controllers due to a failure of one of the manual supply valves is considered negligible and is not modeled. IA supply line valves are all normally open manual valves. The probability of these manual valves transferring closed is much smaller than the failure probability of one of the valves in the SG PORV controllers.	RNP/F-PSA-074, Table 1, Item 65.	Although the PRA model will not generate importance measures for these manual valves, if the instrument air system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these valves, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
14	Failure of Pressure Transmitters that provide input to the PORVs is not modeled. These failures cause an initiating event and are included in the initiating event frequency.	RNP/F-PSA-074, Table 1, Item 69.	Although the PRA model will not generate importance measures for these valves, these transmitters are implicitly modeled since the cause an initiating event. Therefore, if the pressure control system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
15	Multiple service water pump start signals may be generated following any specific initiating event. For simplification of start logic modeling, only the most prevalent signals are modeled. Due to the reliability of the start signals, modeling just the most prevalent start signals is deemed appropriate and the signal reliability underestimation is not a concern.	RNP/F-PSA-074, Table 1, Item 72.	Although the PRA model will not generate importance measures for the components which generate these signals, if the system containing these components is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
16	Failure to isolate non-essential CCW loads on containment isolation phase "A" is not a major CCW flow diversion and does not fail the CCW function. Failing to isolate non-essential CCW loads places excessive demand on a pump, but CCW design starts standby pumps on low discharge pressure.	RNP/F-PSA-074, Table 1, Item 79.	Although the PRA model will not generate importance measures for the isolation valves, if the CCW system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
17	Failure probabilities for diesel auxiliaries and for EDG voltage regulators and K-1 relays are included within diesel generator failure data. Failure of DG starting air, fuel oil, lubrication, cooling systems, EDG voltage regulators and K-1 relays are assumed to be in DG failure data used.	RNP/F-PSA-074, Table 1, Items 80 and 82.	Although the PRA model will not generate importance measures for these components, if the EDG system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
18	Failure of the air dryers within the 24 hour mission time does not result in an acute IA failure. Degradation of air quality would present long-term problems, not in the short term (24 hours).	RNP/F-PSA-074, Table 1, Item 96.	Although the PRA model will not generate importance measures for the air dryers and related components, if the IA system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
19	Failure probability of CS pump coolers is integral to the failure probability for the CS pump. Pump cooler failures are included in the overall pump failure rate.	RNP/F-PSA-074, Table 1, Item 112.	Although the PRA model will not generate importance measures for the coolers, if the CS system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, since they support the function of the pump, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
20	Simultaneous loading of two or more loads onto the Emergency Buses is assumed not to occur. The likelihood of the interposing relays failing in a manner which permits this is negligible.	RNP/F-PSA-074, Table 1, Item 131.	Although the PRA model will not generate importance measures for the interposing relays, if the undervoltage system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these components, since they support the function of the system, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.

Uncertainties and Assumptions Not Addressed by 10CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring			
No.	**Description of Assumption or Source of Uncertainty**	**Source**	**Assessment**
21	Some locked-open manual Fire Water valves are not included in the system model. Regular Fire Water system flow testing confirms correct position of these locked-open valves.	RNP/F-PSA-074, Table 1, Item 137.	Although the PRA model will not generate importance measures for these manual valves, if the fire water system is categorized, appropriate surrogate PRA events will be used to generate importance measures for these valves, since they support the function of the system, or the components will be added to the model. Any impact of the simplified modeling on acceptance criteria for categorization of other components is addressed by the 10CFR 50.69 factor of 3 sensitivity and performance monitoring.
22	Several simplifying assumptions are made regarding the condensate and feedwater system.	RNP/F-PSA-074, Table 1, Items 144, 145, 146 and 147.	The condensate and feedwater system model for the Robinson PSA model is a simplified model containing key components of those systems and their required support systems. It is not intended to provide the capability for detailed examination at the component level. Therefore, if the Feedwater/Condensate system is categorized, appropriate surrogate events may be required for some components, or the system model may need to be updated.

PRA RAI 04 – Very Early Warning Fire Detection Systems (VEWFDS) Utilized in the PRA:

Assumption/Uncertainty No. 4 in Attachment 6 of the LAR states, “[t]he HBRSEP2 Fire PRA assumes Incipient Detection System functions as outlined in NUREG 2180.” The disposition to this uncertainty states, “[t]he current method of crediting Incipient Detection at RNP is similar to NUREG 2180 with more credit for operators to prevent fires based upon actual plant experience and plant procedures.” It is not clear to the NRC staff how much actual plant experience with fires has been collected and what differences exist between NUREG 2180 and the licensee’s approach.

LAR Section 3.2.2 states “[t]he internal Fire PRA model was developed consistent with NUREG/CR-6580 and only utilizes methods previously accepted by the NRC.” However, in a letter dated July 1, 2016, “Retirement of National Fire Protection Association 805 Frequently Asked Question 08-0046 “Incipient Fire Detection Systems” (ADAMS Accession No. ML16167A444), FAQ 08-0046 that was previously accepted by the NRC was retired. In this letter it was requested of licensees to evaluate the impact of the new guidance on their PRA in accordance with their licensing basis. In light of these observations, address the following:

a. Provide justification, such as a sensitivity study, that the use of FAQ 08-0046 VEWFDS methodology, which is not endorsed by the NRC, has no impact on the 10 CFR 50.69 categorization process. If determined that the use of VEWFDS has an impact on the categorization process, provide a detailed description of the method used for crediting Incipient Detection and technical justification for why it is acceptable for use in the 10 CFR 50.69 categorization process.

Duke Energy Response to PRA RAI 04.a.:

The methodology used for crediting incipient detection at HBRSEP2 is NUREG-2180 *Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities,* Final Report Published December 2016.

b. Alternatively, propose a mechanism that ensures the VEWFDS methodology will be updated to be consistent with NUREG-2180, or other current NRC acceptable methodology prior to implementation of the 10 CFR 50.69 categorization process. If this update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review and ensure any findings are closed by using an approved NRC process.

Duke Energy Response to PRA RAI 04.b.:

The HBRSEP2 Fire PRA model was reviewed against NUREG-2180 guidance. There are no differences between the method used at HBRSEP2 and the method described in NUREG-2180. The wording in the LAR with regards to additional credit for operator actions is not implemented in the model. Thus, treatment of incipient detection at HBRSEP2 fully aligns with NUREG-2180 guidance.

PRA RAI 05 – Key Assumptions and Uncertainties that Could Impact the Application:

Section 1.2.10 of RG 1.200 discusses the technical approach in determining the impact of assumptions and sources of uncertainty on the PRA model.

The dispositions presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties state in each case that: “this does not represent a key source of uncertainty and will not be an issue for the 50.69 calculations.” However, in a number of instances there is not enough information provided in the dispositions for the NRC staff to determine whether the uncertainty will not impact 10 CFR 50.69 risk categorization. In light of these observations address the following:

a. Feed and Bleed Success Criteria for loss of secondary heat removal

LAR Attachment 6 (page 48) states that the current PRA MOR success criteria for Feed and Bleed is one high pressure safety injection (HPSI) and two power operated relief valves (PORVs), but the thermal hydraulic analysis concludes only one PORV is required. The disposition states that this could result in certain SSCs in having higher risk significance and therefore is considered conservative. The staff notes that conservative modeling choices can potentially artificially lower other components risk importance values below the safety significance threshold criteria (i.e. masking). In light of these observations, address the following:

i. Provide justification, such as a sensitivity study, that the exclusion of the updated success criteria does not affect any of the SSC risk categorizations.

ii. Alternatively, propose a mechanism to incorporate the updated success criteria into the PRA MOR prior to implementation of the 10 CFR 50.69 categorization program.

Duke Energy Response to PRA RAI 05.a.:

Item 10 of Attachment 6 of the LAR states that the success criteria for feed and bleed is one HPSI pump and two PORVs for loss of all secondary heat removal scenarios, assuming that the loss of heat removal occurs at time T=0. It addresses the uncertainty associated with the assumption that the loss of heat removal occurs at T=0. It recognizes that if the loss of heat removal occurs at a time of approximately 50 minutes or later (when decay heat levels are lower), only a single PORV is required. Since separating loss of all secondary heat removal scenarios that occur prior to 50 minutes from those that occur later than 50 minutes in the PRA model is very complicated (and introduces its own uncertainty), this reasonable and conservative assumption has been used, and is appropriate.

Since this assumption introduces an uncertainty into the application, and it cannot be quantitatively shown that it does not have the potential to impact the acceptance criteria, it is being addressed by the sensitivity study required by section 8.1 of NEI 00-04 and performance monitoring of low safety significant (LSS) components as required by 10 CFR 50.69(e)(3). As discussed in PRA RAI 03, the sensitivity and monitoring program are sufficient to address model uncertainties and assumptions not related to components which were excluded from the model. Since this uncertainty does not result in any components being excluded from the model, this assumption is appropriately addressed by the sensitivity study and performance monitoring program.

b. Operator Action Recovery Dependency Analysis

LAR Attachment 6 (page 48) discusses the floor value applied to HFE combinations. For performing HRA dependency analysis, NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report,” July 2012 (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple HFEs, and refers to NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” April 2005 (ADAMS Accession No. ML051160213) (Table 2-1), which recommends joint human error probability (JHEP) values should not be below 1E-5. Table 4-3 of EPRI TR 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," October 2010, provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Assigning JHEPs that are less than a minimum value should be individually reviewed for timing, cues, etc., to check the dependency between all the operator actions in the cutset.

Assumption/Uncertainty #8 (page 47) provides a statement that the lower bound value of 1E-05 is applied for any individual HFE. However in Assumption/Uncertainty #9, the floor value applied to HFE combinations is not specified. Therefore, provide the following:

i. Clarify the floor value that is applied to HFE combinations in the Robinson IEPRA.

ii. If the floor value is less than 1E-06, provide an estimate of the number of these JHEP values below 1E-6 in the IEPRA, discuss the range of values and confirm that justification is documented for each of these JHEPs.

Duke Energy Response to PRA RAI 05.b.:

A lower bound of 1E-06 was enforced as the limiting JHEP for any two or more HEPs in any one cutset for the RNP Internal Events and Internal Flood models. Even if the dependency combination is calculated to be below 1E-6, this floor is still applied in the quantification of risk results. Therefore, there are no dependency combinations below 1E-6 as this is the absolute floor for any dependency combination no matter the number of single human failure events in the combination.

PRA RAI 06 – Key Assumptions and Uncertainties Subject to Sensitivity Studies:

In LAR Attachment 6, assumptions 1, 2, and 3 address reactor coolant pump seal failure, loss of offsite power frequencies, and fire modelling respectively. Each of these assumptions is dispositioned with,

> *In accordance with NEI 00-04, sensitivity studies will be used to determine whether other conditions might lead to the component being safety significant. The assessment of the uncertainties, therefore, is appropriately addressed by the sensitivity studies required by this risk-informed application.*

NEI 00-04 sensitivity studies in Tables 5-2, 5-3, 5-4, and 5-5 all include human error probabilities, CCF probabilities, and maintenance unavailabilities. The uncertainties in assumptions 1, 2 and 3 are not related to these issues or parameters and therefore the sensitivity studies in the Tables do not resolve the effect of the assumptions. However, each Table also has provision for "[a]ny applicable sensitivity studies identified in the characterization of PRA adequacy" but these PRA specific studies need to be identified. For each Assumption 1, 2, and 3:

a. Describe the applicable sensitivity study that will be undertaken to address each uncertainty or otherwise resolve the effect of the assumption on the categorization process.

Duke Energy Response to PRA RAI 06.a.:

The updated assessment of key sources of uncertainty and assumptions performed in response to RAI-03.b. supersedes the contents of Attachment 6 to the original LAR.

b. Propose a mechanism that ensures that the identified sensitivity studies will be included in the categorization evaluations. This mechanism should also provide an explicit description of the each sensitivity study.

Duke Energy Response to PRA RAI 06.b.:

The updated assessment of key sources of uncertainty and assumptions performed in response to RAI-03.b. supersedes the contents of Attachment 6 to the original LAR. That updated assessment addresses the strategy to address key assumptions and uncertainties for this application.

PRA RAI 07 – SSCs Categorization Based on Other External Hazards:

Section 3.2.4 of the LAR states:

> As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS. All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

The last sentence implies that the assessment has been completed and concludes that all other external hazards will never need evaluation during categorization. The individual plant examination of external events (IPEEE) screening process did not include the additional step illustrated in Figure 5-6 in Section 5.4 of NEI 00-04. Figure 5-6 and its associated text states that an evaluation is performed to determine if there are components being categorized that participate in screened external event scenarios whose failure would result in an unscreened scenario.

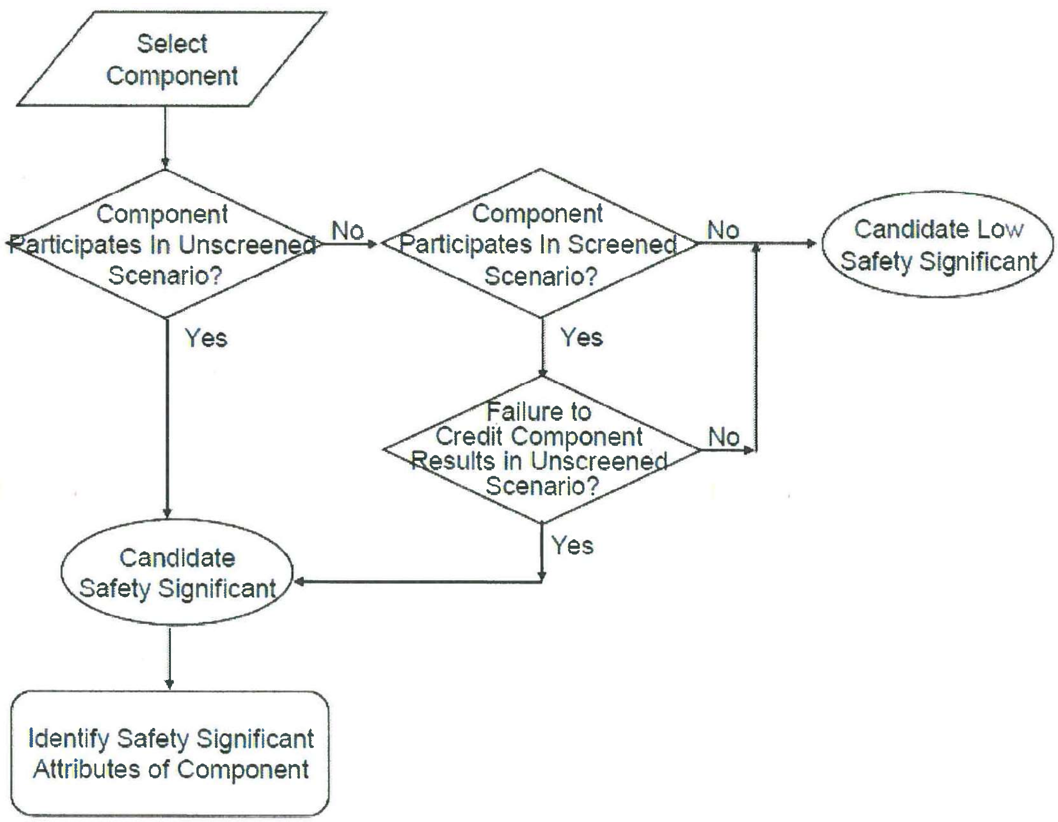
Clarify how the screening criteria in LAR Attachment 5, “Progressive Screening Approach for Addressing External Hazards,” satisfy the guidelines that HSS will be assigned to SSCs whose failure would cause a screened external event scenario to become unscreened.

Duke Energy Response to PRA RAI 07:

The screening criteria in Attachment 5 of the LAR were used to determine those external hazards listed in Attachment 4 of the LAR requiring a PRA model for this application and those

screened from needing a PRA model. The LAR Attachment 5 denotes the screening criteria that determines "screened scenarios" versus "un-screened scenarios".
Per NEI 00-04 the external hazard assessment is required for each SSC categorization. As such, each SSC being categorized will be assessed in accordance with NEI 00-04 Figure 5-6 for the external hazards listed in Attachment 4 of the LAR. If the failure of the SSC results in the screening criterion from Attachment 5 not being met, then the scenario would become unscreened and the SSC would become candidate High Safety Significant. NEI 00-04 Figure 5-6 is shown below for reference.

Figure 5-6
OTHER EXTERNAL HAZARDS



PRA RAI 08 – Incorporation of FLEX into the PRA Model(s):

The NRC memorandum dated May 30, 2017, "Assessment of The Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200. The LAR does not state whether or not the licensee has incorporated FLEX mitigating strategies and associated equipment into the PRA models at Robinson.

Provide the following information separately for internal events PRA, external hazard PRAs, and external hazard screening as appropriate:

a. If FLEX mitigating strategies and associated equipment have not been incorporated into the base PRA and the external hazard evaluations, confirm that FLEX equipment is not modelled.

Duke Energy Response to PRA RAI 08.a.:

FLEX mitigating strategies and associated equipment have not been incorporated into the current Robinson PRA models of record for internal events or external hazards. There is no FLEX equipment in the current model.

b. If FLEX mitigating strategies and associated equipment have been incorporated into the base PRA and the external hazard evaluations but do not impact the categorization process, summarize the evaluation supporting the conclusion that there is no impact on categorization.

Duke Energy Response to PRA RAI 08.b.:

The scenario described in RAI 08.b. is not applicable to HBRSEP2.

c. If FLEX mitigating strategies and associated equipment have been incorporated into the base PRA and the external hazard evaluations and do impact categorization, provide the following information:

   i. A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application.

   ii. A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee's mitigating strategies (i.e., FLEX). The discussion should include a justification explaining the rational for parameter values, and whether the uncertainties associated with the parameter values are considered in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200.

   iii. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

      (1) A summary of how the licensee evaluated the impact of the plant-specific HEPs and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA standard.

(2) Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA standard.

(3) If the licensee's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

iv. The ASME/ANS RA-Sa-2009 Standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

(1) Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria are satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences, OR

(2) Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization program.

Duke Energy Response to PRA RAI 08.c.:

Items i.-iv. in RAI 08.c. are not applicable to HBRSEP2.

PRA RAI 09 – Proposed License Condition:

The guidance in NEI 00-04 allows licensees to implement different approaches, depending on the scope of their PRA (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a seismic PRA is used). RG 1.201, Revision 1 states that "as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement §50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non- PRA methods used in the licensee's categorization approach."

Section 2.3 of the LAR Supplement proposed the following License Condition:

Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-

4 structures, systems, and components (SSCs) specified in the license amendment request dated April 5, 2018.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The proposed license condition does not explicitly address the PRA and non-PRA approaches that were used. Provide a license condition that explicitly address the approaches, e.g.:

Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

Note that if implementation items are identified, the license condition may need to be expanded to address them.

Duke Energy Response to PRA RAI 09:

Duke Energy proposes the following license condition, which is also reflected in the HBRSEP2 Operating License markup in Attachment 2 of this submittal.

Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. [XXX] dated [DATE].

Duke Energy will complete the implementation items list in Attachment 1 of Duke Energy letter to the NRC dated November 13, 2018 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

Serial: RA-18-0193

H.B. Robinson Steam Electric Plant, Unit 2
Docket No. 50-261 / Renewed License No. DPR-23

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors”

Attachment 1

HBRSEP2 50.69 PRA Implementation Items

The table below identifies the items that are required to be completed prior to implementation of 10 CFR 50.69 at H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2). The issues identified below will be addressed and any associated changes made, focused scope peer reviews performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and findings resolved and reflected in the PRA of record prior to implementation of 10 CFR 50.69.

Robinson 50.69 PRA Implementation Items	
Description	Resolution
i. The HBRSEP2 internal flood model does not account for generic human induced flooding data as described in response to RAI 1.b. in Duke letter dated November 13, 2018. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing 50.69.	Duke Energy will update the HBRSEP2 internal flood model to account for generic human induced flooding events using an industry accepted methodology described in the response to RAI 1.b. in Duke letter dated November 13, 2018.
ii. Human Failure Events (HFEs) related to isolating a ruptured SG following a thermally induced steam generator tube rupture (SGTR) are not represented in the internal events model as described in response to RAI 1.a. in Duke letter dated November 13, 2018. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing 50.69.	Duke Energy will update the RNP internal events model to include these operator actions.

Serial: RA-18-0193

H.B. Robinson Steam Electric Plant, Unit 2
Docket No. 50-261 / Renewed License No. DPR-23

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors”

Attachment 2

Markup of Proposed Renewed Facility Operating License

No changes on this page.

APPENDIX B

ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. DPR-23

Duke Energy Progress, LLC. (the term licensee in Appendix B refers to Duke Energy Progress, LLC.) shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
176	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated September 10, 1997, and October 13, 1997, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
219	Upon implementation of the amendment adopting TSTF–448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by TS 5.5.17.c.(i), the assessment of CRE habitability as required by TS 5.5.17.c.(ii), and the measurement of CRE pressure as required by TS 5.5.17.d, shall be considered met. Following implementation: (a) The first performance of TS 5.5.17.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from January 27,2003, the	This amendment is effective immediately and shall be implemented as specified

date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, TS 5.5.17.c.(ii), shall be within the next 9 months.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.17.d, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test.

INSERT 1

INSERT 1

Amendment Number	Additional Conditions	Implementation Date
[NUMBER]	Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and external flood; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. [XXX] dated [DATE]. Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated November 13, 2018 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).	Upon implementation of Amendment No. [XXX].